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BECHTEL SAIC COMPANY LLC

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October 2004

Clad Degradation – Summary and Abstraction for LA

Prepared for:
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OCRWM	MODEL SIGNATURE PAGE/CHANGE HISTORY	Page iii 1. Total Pages: 58

2. Type of Mathematical Model

☐ Process Model ☒ Abstraction Model ☐ System Model

Describe Intended Use of Model

The cladding degradation model is used in TSPA-LA to describe how the commercial spent nuclear fuel cladding degrades in the repository during the postclosure period.

3. Title

Clad Degradation – Summary and Abstraction for LA

4. DI (including Rev. No., if applicable):

ANL-WIS-MD-000021 REV 01

5. Total Appendices	6. No. of Pages in Each Appendix
None	N/A

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13. Remarks

Change History

14. Revision No.	15. Description of Change
00	Initial Issue.
01	Complete revision to address changes due to regulatory integration. Change bars were not used because the changes were too extensive to track.

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EXECUTIVE SUMMARY

The purpose of this model report is to develop the summary cladding degradation abstraction that will be used in the Total System Performance Assessment for the License Application (TSPA-LA). Most civilian commercial nuclear fuel is encased in Zircaloy cladding. The model addressed in this report is intended to describe the postulated condition of commercial Zircaloy-clad fuel as a function of postclosure time after it is placed in the repository. Earlier total system performance assessments analyzed the waste form as exposed UO_2, which was available for degradation at the intrinsic dissolution rate. Water in the waste package quickly became saturated with many of the radionuclides, limiting their release rate. In the total system performance assessments for the Viability Assessment and the Site Recommendation, cladding was analyzed as part of the waste form, limiting the amount of fuel available at any time for degradation.

The current model is divided into two stages. The first considers predisposal rod failures (most of which occur during reactor operation and associated activities) and postdisposal mechanical failure (from static loading of rocks) as mechanisms for perforating the cladding. Other fuel failure mechanisms including those caused by handling or transportation have been screened out (excluded) or are treated elsewhere. All stainless-steel-clad fuel, which makes up a small percentage of the overall amount of fuel to be stored, is modeled as failed upon placement in the waste packages. The second stage of the degradation model is the splitting of the cladding from the reaction of water or moist air and UO_2. The splitting has been observed to be rapid in comparison to the total system performance assessment time steps and is modeled to be instantaneous. After the cladding splits, the rind buildup inside the cladding widens the split, increasing the diffusion area from the fuel rind to the waste package interior.

This model report summarizes the component models, developed for the two stages noted above, that are used as inputs to TSPA-LA. The model concludes that less than two percent of the fuel, including all of the stainless-steel clad fuel, received at the repository is failed (perforated) upon receipt at the repository. All failed fuel is assumed to axially split upon waste package failure exposing the fuel to oxidation from the in-package environment. TSPA-LA then calculates the release of radionuclides from the exposed volume of oxidized fuel.

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TABLES

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ACRONYMS

ANL	Argonne National Laboratory
ASTM	American Society for Testing and Materials
BWR	boiling water reactor
CSNF	Commercial Spent Nuclear Fuel
DOE	U.S. Department of Energy
DTN	data tracking number
EPRI	Electric Power Research Institute
FEPs	features, events, and processes
LWR	light water reactor
NRC	U.S. Nuclear Regulatory Commission
PWR	pressurized water reactor
SNF	spent nuclear fuel
TER	Technical Error Report
TIC	Technical Information Center
TSPA	total system performance assessment
TSPA-LA	Total System Performance Assessment for the License Application
TSPA-SR	Total System Performance Assessment for the Site Recommendation

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1. PURPOSE

The purpose of this model report is to develop the summary cladding degradation abstraction to be used in the Total System Performance Assessment for the License Application (TSPA-LA). The scope of this model is to evaluate commercial spent nuclear fuel (CSNF) cladding under the TSPA-LA repository design. This document was prepared in accordance with *Technical Work Plan for: Regulatory Integration Modeling and Analysis of the Waste Form and Waste Package* (BSC 2004 [DIRS 171583]).

Most civilian commercial nuclear fuel is encased in Zircaloy cladding. Zircaloy is high-purity zirconium alloyed with tin with small additions of nickel and chromium, with and without small additions of iron. The term “Zircaloy” is also inclusive of newer alloys that have a lower tin content or have small additions of niobium. The model has been developed to describe cladding degradation from the expected failure modes. These modes include failure before receipt at the repository (due to reactor operation impacts, during spent fuel storage in pool and dry storage, and handling) and degradation in the repository (mechanical failure). It is expected that unfailed (not breached or perforated) Zircaloy cladding will remain intact for at least 10,000 years, the prescribed regulatory period. For those fuel rods that are modeled to breach, the cladding is conservatively modeled to be axially split open due to strain induced by fuel oxidation and swelling or by growth of the oxide layer on the inside of the cladding. This does not occur until the waste package is breached. Stainless-steel-clad fuel is also addressed in this model report. However, all stainless steel cladding is assumed to be failed prior to emplacement.

There are constraints, caveats, and limitations to this cladding degradation model. This model is based on commercial water reactor fuel with Zircaloy cladding. The model discussed herein does not apply to fuel from a commercial gas cooled reactor (Fort Saint Vrain), which is included in the inventory of U.S. Department of Energy (DOE) fuels. DOE fuels are not addressed in this report. Naval Nuclear Propulsion Program cladding/SNF performance is discussed in *Naval Nuclear Propulsion Program Technical Support Document for the License Application,* which is a classified document.

Commercial spent nuclear fuel reliability from reactor operation is determined for both pressurized water reactors (PWRs) and boiling water reactors (BWRs). This model is also limited to fuel exposed to normal operation and anticipated operational occurrences (i.e., events that are anticipated to occur within a reactor lifetime) and is not applicable to fuel that has been exposed to severe accidents. It is also limited to a repository design in which the fuel rod temperature is always less than 350°C during postclosure. Fuel burnup projections have been limited to the current commercial reactor licensing environment that restricts fuel enrichment, oxide coating thickness, and rod plenum pressures. Input uncertainties are discussed in Sections 4.1 and 6, while model uncertainties are summarized in Section 6.5. The information provided in this model will be used in evaluating the postclosure performance of the repository in relation to waste form degradation.

This model report complements the work reported in *Clad Degradation – Summary and Abstraction* (CRWMS M&O 2001 [DIRS 151662]), including additional validation of the cladding degradation models. This referenced report contains the cladding abstraction used in the Total System Performance Assessment for the Site Recommendation (TSPA-SR) model. *Clad*

Degradation – Summary and Abstraction (CRWMS M&O 2001 [DIRS 151662]) is not being superseded or revised because it contains a statistical creep analysis that is referenced for excluded FEPs. These two earlier reports were written as analysis reports instead of as model reports, and did not contain extensive validation and alternative conceptual model sections. This new report expands the discussions of validation, alternative conceptual model, and uncertainties.

2. QUALITY ASSURANCE

The Quality Assurance program applies to the development of this document because this model will be part of the TSPA-LA safety analysis. This document was prepared in accordance with *Technical Work Plan for: Regulatory Integration Modeling and Analysis of the Waste Form and Waste Package* (BSC 2004 [DIRS 171583]), which directs the work identified in work package ARTM03. The technical work plan was prepared in accordance with AP-2.27Q, *Planning for Science Activities*. All input data for the model document are identified and tracked in accordance with AP-3.15Q, *Managing Technical Product Inputs*. This report indirectly impacts structures, systems, or components classified in accordance with AP-2.22Q, *Classification Criteria and Maintenance of the Monitored Geologic Repository Q-List* as Safety Category (SC) in the *Q-List* (BSC 2004 [DIRS 168361]). The technical work plan contains the Process Control Evaluation used to evaluate the control of electronic management of data (BSC 2004 [DIRS 171583], Appendix A) during modeling and documentation activities. This evaluation determined that the methods identified in the implementing procedures are adequate. Data submittal was consistent with AP-SIII.3Q, *Submittal and Incorporation of Data to the Technical Data Management System.* No deviations from these methods were performed. As directed in the technical work plan, this document was prepared in accordance with AP-SIII.10Q, *Models,* and reviewed in accordance with AP-2.14Q*, Review of Technical Products and Data*.

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3. USE OF SOFTWARE

No software is used for modeling in this model report. This model report was documented using only commercially available software (Microsoft Word 97, SR2) for word processing, which is exempt from qualification requirements in accordance with LP-SI.11Q, *Software Management*. There were no additional applications (routines or macros) developed for documentation using this commercial software.

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4. INPUTS

4.1 DATA AND PARAMETERS

The project data is presented by data tracking number (DTN) or accession number, while nonproject technical information is presented by Technical Information Center (TIC) catalog numbers in this section. Both data and technical information are summarized in Table 4-1. These sources are not used to validate the model.

1. TIC: 242976 (Roberts et al. 1990 [DIRS 107105], pp. 766 to 767)

   The density of schoepite (Table 4-1, Item 2) is used in Section 6.2.4 to calculate the split geometry as the pellet corrodes. The split geometry is directly related to the change in volume from unreacted UO_2 to schoepite. This source is the *Encyclopedia of Minerals* and values used are Established Fact.

2. DTN: LL010902212241.026 [DIRS 163089]

   The porosity of schoepite (Table 4-1, Item 3) is used in Section 6.2.4 to calculate the split geometry as the pellet corrodes. The three values of porosity for the schoepite rind (0.05, 0.15, and 0.30) represent schoepite exposed to two different flow rates of water and the presence of water vapor.

3. TIC: 243741 (Lide and Frederikse 1997 [DIRS 103178], p. 4-94)

   The theoretical density and molecular weight of UO_2 (Table 4-1, Item 4) is used in Section 6.2.4 to calculate the split geometry as the pellet corrodes. This source is *CRC Handbook of Chemistry and Physics* and values used are Established Fact.

4. TIC: 245486 (Preble et al. 1993 [DIRS 107407]).

   The fuel pellet diameter, number of rods in an assembly, and active rod length for a typical (Westinghouse) 17×17 PWR assembly (Table A.1 of the reference) is used in Section 6.2.4 to calculate the cladding split geometry as the pellet corrodes. This data source was used because the majority of the SNF proposed for inclusion in the Yucca Mountain Repository will come from this type of PWR. Thus, this source contains the relevant information needed for this evaluation. This input source is qualified for its intended use within this technical product (report) per AP-SIII.10Q, Section 5.2.1(k), as corroborating data are available. The same information (e.g., pellet diameter, rods per assembly, and active fuel rod length) is found in a report by Roddy et al. (1985 [DIRS 120630]), which was independently developed by Oak Ridge National Laboratory using database information acquired directly from fuel rod manufacturers.

5. TIC: 246541 (S. Cohen & Associates 1999 [DIRS 135910], p. 6-11, Table 6.2)

S. Cohen & Associates performed a study of the effectiveness of fuel rod cladding as an engineered barrier in the repository. The U.S. Environmental Protection Agency (EPA) sponsored this study. As part of the study, S. Cohen & Associates evaluated as-received SNF cladding failures. Failures from reactor operation, pool storage, dry storage, rod consolidation, and other handling failures were reviewed (S. Cohen & Associates 1999 [DIRS 135910], p. 7-1). Because the SNF that is proposed for the Yucca Mountain Repository will have come from similar reactors and will have been treated to the same storage processes, the data from this investigation are directly applicable as TSPA-LA input. The authors conclude that the expected value of the parameter for failed fuel, as-received, is 0.1% with the range being 0.01% to 1%, based on historical data. The lowest value quoted in their summary table was 0.04%; however, the referenced utility data go down to 0.01% and lower.

The investigators in these studies are recognized national experts in the field and the U.S. EPA contracted for these data to support their efforts to develop 40 CFR 197 regulations specifically for the Yucca Mountain Repository. This technical information is appropriate to use in this model report because it is an independent evaluation of as-received fuel failures. Thus, the data are considered qualified for use within this technical product based upon AP-SIII.10Q, Section 5.2.1(k), specifically in regard to the qualifications of the personnel and organization generating the data.

Table 4-1. Input Values to Clad Degradation Model

Item	Input name	Source and Location	Value
1	Theoretical Density of Schoepite	Roberts et al 1990 [DIRS 107105] Schoepite properties	4.83 (calculated) g/cm^3
2	Porosity of Schoepite, fraction	DTN: LL010902212241.026 [DIRS 163089], Rows 1, 2, 3	0.05, 0.15, 0.30
3	Density and molecular weight of UO_2	Lide and Frederikse 1997 [DIRS 103178], pp. 4-94	10.97 g/cm^3, 270 g/mol
4	Fuel rods/assembly	Preble et al [DIRS 107407] Table A.1	264
5	Active fuel rod length, cm	Preble et al [DIRS 107407] Table A.1	144 in. (3.66 m)
6	Fuel pellet diameter	Preble et al [DIRS 107407] Table A.1	0.3225 in. (0.819 cm)
7	Distribution of failed cladding, as-received	S. Cohen & Associates 1999 [DIRS 135910], p. 6-11, Table 6.2, Row 1	0.1% expected value, range 0.01% to 1%

4.2 CRITERIA

The following acceptance criteria, or portions therefrom, from *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274]) were identified as applicable to this model report. A detailed description is available in the referenced document. Criterion 3 is an addition to those identified in the technical work plan (BSC 2004 [DIRS 171583]). Criterion 3 (taken from NRC 2003 [DIRS 163274], Section 2.2.1.3.2.3) was added as the mechanical disruption of engineered barriers (i.e., the cladding as part of the waste form) is an important mechanism for cladding failure. The detail on how these criteria have been addressed is provided in Section 8.3.

1. System Description and Demonstration of Multiple Barriers, Acceptance Criteria (NRC 2003 [DIRS 163274], Section 2.2.1.1.3). The barriers are to be adequately identified and described (including time periods, uncertainty). The technical basis for the barrier capability is adequately presented, commensurate with the importance of the barrier.

2. Degradation of Engineered Barriers, Acceptance Criteria (NRC 2003 [DIRS 163274], Section 2.2.1.3.1.3). System description and model integration for the degradation of the barriers are adequate. Model adequately addresses design features, physical phenomena, and couplings and uses appropriate assumptions. Boundary and initial conditions used are propagated appropriately and consistently. Data are sufficient for model justification. Data uncertainty is characterized and propagated through model abstraction.

3. Mechanical Disruption of Engineered Barriers, Acceptance Criteria (NRC 2003 [DIRS 163274], Section 2.2.1.3.2.3). System description and model integration for the mechanical disruption of the barriers are adequate. Data on mechanical disruptions are sufficient for model justification. Data uncertainty for mechanical disruptions are characterized and propagated through model abstraction.

4. Radionuclide Release Rates and Solubility Limits, Acceptance Criteria (NRC 2003 [DIRS 163274], Section 2.2.1.3.4.3). System description and model integration are adequate. Data are sufficient for model justification. Data uncertainty is characterized and propagated through the model abstraction. Model uncertainty is characterized and propagated through model abstraction. Model abstraction output is supported by objective comparisons.

4.3 CODES AND STANDARDS

American Society for Testing and Materials (ASTM) Standard C 1174-97 [DIRS 105725], *Standard Practice for Prediction of the Long-Term Behavior of Materials, Including Waste Forms, Used in Engineered Barrier Systems (EBS) for Geological Disposal of High-Level Radioactive Waste*, is used to support the degradation model development methodology, categorize the model developed with respect to its usage for long-term TSPA-LA, and relate the information/data used to develop the model to the requirements of the standard.

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5. ASSUMPTIONS

5.1 STAINLESS-STEEL-CLAD FUEL

Assumption: It is assumed in this model report that there is a finite probability that the contents of any transportation cask arriving at the Yucca Mountain site that contains only stainless-steel-clad assemblies will be loaded into three waste packages. This would result in about 3.5% of the total waste packages that would contain stainless-steel-clad assemblies. This assumption is necessary because it addresses an absence of direct confirming evidence of how the waste packages are to be loaded.

Rationale: Because there are no reactors currently operating that use stainless steel cladding, the total number of fuel assemblies with this cladding is fixed. EPRI (1996 [DIRS 160968]) gives a value of 2,118 such assemblies, while S. Cohen & Associates (1999 [DIRS 135910]) gives a value of 2,179. The larger value is used herein. These assemblies could be loaded into about 86 waste packages (Section 6.2.2). Since the total number of CSNF waste packages is 7,472 (BSC 2004 [DIRS 170022]), the percentage of waste packages that could contain stainless-steel-clad fuel rods is about 1.15%. It is conceivable that a delivery of stainless steel assemblies at the repository in a cask containing, for example, 21 or more PWR assemblies, could be distributed into three waste packages. This is possible either due to the fact that all of the stainless steel could not be accommodated in one waste package or that that the stainless steel assemblies, which are cooler, could be used to fill up waste packages containing hotter, fresher fuel assemblies. Thus, three times 1.15% would yield a potential number of 3.45% (which is rounded up to 3.5%), indicating that about 3.5% of the total waste packages might contain SNF with stainless steel cladding.

Confirmation Status: The product of waste packages containing stainless-steel-clad fuel rods and fraction of stainless steel in each waste package is constant (i.e., there is a fixed amount of stainless-steel-clad fuel rods). Thus, the assumption of a value of 3.5% of waste packages that contain stainless steel assemblies is reasonable and slight variations in this distribution are not expected to have significant impact on dose to the public. Thus, it does not require confirmation.

Use in the Model: This assumption is used in Sections 6.2.2 and 7.4.2.

5.2 ROCK OVERBURDEN

Assumption: It is assumed in this model report that the cladding can fail from mechanical (static) loading when a minimum of 20% (with a uniform distribution of 20% to 50%) of the patches on both the waste package and drip shield surface are corroded open. At the 50% level of corroded patches, failure of the waste package and hence the cladding by static overburden is likely. The lower value of 20% is uncertain, but reasonable and conservative as a lower bound (patches are rectangular areas established on the waste package and drip shield surfaces to facilitate corrosion calculations). Section 6.6 of *Seismic Consequence Abstraction* (BSC 2004 [DIRS 169183]) states that no rockfall is expected on waste packages as a result of protection provided by the drip shields for the first 10,000 years. Thus, the 20% value is conservative. It is further assumed that the fraction of failed fuel rods increases linearly with the number of waste package patch openings and the fuel is 100% failed when 50% of the waste package patches are open. This assumption is necessary because there is an absence of direct confirming data or evidence that would allow a firm determination of when such a event might be postulated to occur.

Rationale: Section 6.2 of *Breakage of Commercial Spent Nuclear Fuel Cladding by Mechanical Loading* (CRWMS M&O 1999 [DIRS 136105]) concluded that static loading from rock overburden onto the fuel assemblies would fail the cladding. The rocks are postulated to fall onto the waste package after the drip shield fails and well before the waste package fails. The surface of the waste package is divided into patches in the TSPA-SR model (CRWMS M&O 2000 [DIRS 153246]) as well as the TSPA-LA model. It is postulated that corrosion causes these patches to open. The location of corroded patches is randomly distributed on the waste package surface. As the patches open on the waste package, the rocks slowly load the cladding. The cladding failure starts when sufficient waste package patches are open to permit rock pressure to start static loading the assemblies. There is uncertainty in the prediction of when the corroded waste package starts to buckle; thus, it is assumed that cladding starts to fail when 20 to 50% of the waste package patches are open.

Confirmation Status: Since this assumption is reasonable and conservative, it does not require confirmation.

Use on the Model: This assumption is used in Section 6.2.3.

5.3 INSTANT SPLITTING

Assumption: It is assumed in this model report that the cladding instantly splits along its length when the cladding is perforated if the waste package has failed. This assumption is necessary because it addressed an absence of direct confirming evidence of how cladding might split when the waste package fails. Note that initially intact cladding can fail due to rock overburden as noted in Section 5.2.

Rationale: The basis for this assumption is past experience with dry splitting of Zircaloy-clad LWR fuel rods due to spent fuel oxidation, wet splitting of BWR rods, and wet splitting of PWR rods in the Argonne National Laboratory (ANL) tests. The former tests are described in Section 6.3.4. For these tests, splitting was observed to occur rapidly after an incubation time was achieved. Note, however, that the incubation time is a function of temperature and perhaps relative humidity.

ANL has performed fuel degradation tests with two intentionally failed fuel segments in humid air at 175°C (Cunnane et al. 2003 [DIRS 162406]). These conditions are more severe than the repository conditions. The cladding on both of these test samples split axially in less than two years. Thus, it is conservatively assumed that BWR and PWR fuel rods will split instantly after waste package breach if they contain perforations at the time of emplacement.

It is also assumed that the rods clad with stainless steel will also split once the waste package has failed. This is based on the conservative assumption that all of the stainless-steel-clad fuel rods will contain perforations. Stainless steel is more susceptible to general corrosion and stress corrosion cracking than Zircaloy and the Alloy 22 waste package material. Thus, it is unlikely that credit for the stainless steel cladding would be defendable.

Confirmation Status: Since this assumption is conservative and consistent with experimental observations, it does not require confirmation.

Use on the Model: This assumption is used in Sections 6.1.2, 6.2.2, and 6.2.4.

6. MODEL DISCUSSION

This section contains the development of a cladding degradation model. It starts with a discussion of cladding as a barrier. The various parts of the cladding degradation abstraction model are then discussed. Alternative conceptual models are presented and a summary of included features, events, and processes (FEPs) is presented. A discussion of uncertainties completes this section. The following list contains the source for corroborating information used to develop the model:

1.* FY01 Supplemental Science and Performance Analyses, Volume 2: Performance Analyses (BSC 2001 [DIRS 154659]), TSPA with alternative cladding degradation model

2. *Risk Information to Support Prioritization of Performance Assessment Models* (BSC 2003 [DIRS 168796]), analysis with naval fuels

3. *Seismic Consequences Abstraction* (BSC 2004 [DIRS 169183]), seismic event occurrences, rockfall impact

4. *Total System Performance Assessment - 1995: An Evaluation of the Potential Yucca Mountain Repository* (CRWMS M&O 1995 [DIRS 100198]), TSPA excluding cladding

5. *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998 [DIRS 100362]), inclusion of cladding in TSPA

6.* *Breakage of Commercial Spent Nuclear Fuel Cladding by Mechanical Loading* (CRWMS M&O 1999 [DIRS 136105]), mechanical failure of cladding

7.* *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000 [DIRS 153246]), sensitivity studies

8.* *Clad Degradation - Dry Unzipping* (CRWMS M&O 2000 [DIRS 149230]), dry oxidation splitting model

9.* *Clad Degradation – Summary and Abstraction* (CRWMS M&O 2001 [DIRS 151662]), fuel as-received failures

10. *Yucca Mountain Project Report, Waste Form Testing Work*, Argonne National Laboratory, Argonne, Illinois, Cunnane et al 2003 [DIRS 162406], cladding axial splitting

11. *Evaluation of the Candidate High-Level Radioactive Waste Repository at Yucca Mountain Using Total System Performance Assessment, Phase 5* (EPRI 2000 [DIRS 154149]), inclusion of cladding in TSPA

12. *Effectiveness of Fuel Rod Cladding as an Engineered Barrier in the Yucca Mountain Repository* (S. Cohen & Associates 1999 [DIRS 135910]), fuel reliability

13. "Sensitivity Studies of the Effect of Cladding Degradation on TSPA Results" (Siegmann and Devonec 2002 [DIRS 160787]), sensitivity studies

14.* Performance Assessment of U.S. Department of Energy Spent Fuels in Support of Site Recommendation (BSC 2001 [DIRS 152059]), comparison of naval to CSNF.

15. *Pitting Model for Zirconium-Alloyed Cladding*, BSC 2004 [170043], pitting of Zircaloy.

Note that the documents marked with asterisks have been cancelled, but are still relevant to this report. The cancelled documents were originally prepared during the site characterization process and are archived but are not kept up to date. They contain useful historical information. They are all qualified for their intended use within this report since they meet the requirements of AP-SIII.10Q Section, 5.2.1(k) specifically because there are prior uses of the data and there are corroborating data available.

This model report describes the cladding degradation process that is to be incorporated into the TSPA-LA model as an abstraction. Cladding degradation consists of two stages, cladding failure (perforation) followed by axial splitting of the cladding. "Axial splitting" was referred to as "unzipping" in earlier DOE publications, but the terminology has been changed to "axial splitting" because this term is widely used in the nuclear industry. Cladding failure is the formation of cracks or holes in the cladding from various sources. The sources of cladding failure that are abstracted into the TSPA-LA model are failures during reactor operation and subsequent storage (including other operations before being received at the repository) and mechanical failure (breaking of the cladding). Unfailed (not perforated) cladding is expected to remain intact for the 10,000-year regulatory period since no credible mechanism has been identified that could cause perforation of the cladding in the inert and benign conditions existing inside an unfailed waste package. Cladding failure permits the fuel inside the cladding to begin to react with water or moist air and potentially leads to the splitting of the cladding. In the splitting stage, the cladding is axially split open by the formation of secondary mineral phases (higher oxides of UO_2, commonly called rind on the fuel, or by fuel-side oxidation of the Zircaloy), making the radionuclides available for release. The split widens as the rind forms and radionuclides diffuse through the split opening. The various components of the abstraction are discussed below.

Fuel damage resulting from handling at the repository surface facility is considered to be negligible. Fuel damage from post-reactor handling operations, other than consolidation, was estimated to be 0.0003% (S. Cohen & Associates 1999 [DIRS 135910], p. 7-1) to 0.0005% (CRWMS M&O 2001 [DIRS 151662], Section 6.4.1). These low failure rates show that handling errors do not make a significant contribution to fuel failure rates.

Naval Nuclear Propulsion Program cladding/SNF performance is discussed in the *Naval Nuclear Propulsion Program Technical Support Document for the License Application* which is a classified document. Based upon the results from a sensitivity study, (BSC 2001 [DIRS

152059]), waste packages containing naval fuel are conservatively modeled as containing CSNF. There are 300 waste packages containing naval SNF compared to the commercial SNF inventory of approximately 7,500 waste packages. A comparison with an equivalent amount of Zircaloy-clad CSNF indicates that the total dose from the TSPA simulation, using the commercial-fuel equivalent, is more than four orders of magnitude higher than the total dose from the source-term simulation for naval SNF (BSC 2001 [DIRS 152059], p. 36 and Figure 6.1-2). Therefore, it is conservative to model naval SNF as CSNF.

6.1 CLADDING AS A BARRIER

Per 10 CFR 63.2 [DIRS 156605], the waste form is one of the recognized barriers that significantly decrease the mobility of radionuclides. That integrity, in large part, depends on the material that encases the actual fuel, the cladding. Cladding prevents the environment from interacting with the fuel pellets by limiting the water or moist air from contacting and interacting with the fuel pellets and reduces radionuclide transport by reducing the diffusion area.

6.1.1 Barrier Capability

All CSNF currently in use in the United States (both PWR and BWR fuel) consists of assemblies of Zircaloy tubing filled with UO_2 fuel pellets. Zircaloy is used because it is resistant to corrosion and has favorable neutronic properties. It has good mechanical properties that enable it to be made into long thin-walled tubing. The first function of the cladding, as a barrier, is to prevent water or moist air from contacting the fuel pellets. The cladding is completely sealed and only fuel with failed cladding can interact with water or moist air. Some cladding could be damaged before it is received at the repository. The distribution of failed cladding as-received is discussed in Section 4.1. The expected value is 0.1%. After repository closure, the cladding could fail due to mechanical causes. Mechanical failure could occur from events associated with seismic activities such as rockfall or rock overburden. The seismic events can occur at any time and are being addressed in a specific seismic TSPA calculation as described in *Seismic Consequences Abstraction* (BSC 2004 [DIRS 169183]). Cladding failure from rock overburden occurs as the waste package and drip shield become severely degraded and no longer protect the fuel from the static loading of the collapsed drift, which does not occur in the 10,000-year regulatory period (BSC 2004 [DIRS 169183], Section 6.6).

After the cladding fails, water or moist air can enter the fuel rod and interact with the fuel pellets. This interaction releases the soluble fission products and actinides that are bound up in the pellet and also increases the volume of the pellet by forming less dense phases. Corrosion of the interior of the cladding is also possible. The increases in volume of the pellet or cladding can axially split the cladding, permitting greater fuel surface area to be exposed to the environment. Failed cladding still somewhat retards radionuclide mobilization by limiting the amount of water or moist air that can contact the pellet and reduces radionuclide transport by reducing the diffusion area for radionuclide release. In the model for TSPA-LA, no credit is taken for failed cladding limiting the amount of water or moist air contacting the fuel but credit is taken for cladding reducing the release of radionuclides because of the limited diffusion area through the split cladding (Section 6.2.4).

6.1.2 Barrier Time Period of Function

Cladding deterioration is modeled in two steps, failure (perforation) and splitting. Failures may occur as a result of seismic events or from mechanical failure due to the static load of rock overburden. Seismic failures and resultant failures may occur at any time. However, no significant cladding failures are predicted to occur after emplacement until both the waste package and drip shield have failed.

Splitting of previously perforated cladding starts when the waste package fails, permitting water or moist air into the package. After waste package failure, any previously failed cladding is assumed to split immediately. Uncertainty exists regarding the time period required for splitting to occur. ANL (Cunnane et al. 2003 [DIRS 162406]), has performed fuel degradation tests with two intentionally failed fuel segments in humid air at 175°C. Both of these fuel segments split in less than two years, a very rapid time scale in terms of the TSPA-LA model. Although the temperature of the waste package internals are not expected to be this high after waste package breach (except for the case of postulated early waste package failure when little moisture is present), splitting may also occur at lower temperatures, perhaps at a lower rate. Therefore, in the TSPA-LA model, the cladding is assumed to instantly split (Assumption 5.3) when both waste package and cladding perforation exists, leaving the fuel pellets exposed for corrosion.

6.1.3 Barrier Uncertainty

The uncertainties associated with the models for failing the cladding are described in this model report and included in the TSPA-LA model. The uncertainty in the fraction of fuel that is failed as-received at the repository is based on observed fuel reliability from reactor operation and conservative predictions for other activities after removal from the reactor. The uncertainties in the splitting of the cladding model are large. Thus, a conservative approach was taken for the splitting of the cladding. Since a conservative model (instantaneous splitting) is used, no uncertainty is used in this portion of the TSPA-LA model.

6.1.4 Barrier Consistency with TSPA

Barrier performance in terms of preventing water or moist air from contacting the fuel is modeled in the TSPA-LA model. As-received failures and failures from seismic and rock overburden are included in the TSPA-LA model. No credit is included in the TSPA-LA for failed cladding reducing or slowing down the fuel pellets from reacting with the environment after cladding failure. In the TSPA-LA model, the fuel pellets corrode at the forward rate of reaction as if they were bare pellets after cladding failure. This is conservative since any oxygen starvation or moisture limitations would reduce the corrosion rate. The failed cladding does reduce the rate at which the radionuclides would be released. Diffusion of radionuclides across the split opening is modeled. The full family of sensitivity studies shows that TSPA results are not sensitive to the splitting model. (See, for example, Siegmann and Devonec 2002 [DIRS 160787].)

6.2 COMMERCIAL FUEL ROD CLADDING DEGRADATION ABSTRACTION MODEL

ASTM C1174-97 [DIRS 105725] establishes guidance for the development of models used for the degradation of engineered barrier system materials based on physical laws, conceptual models, and relatively short-term, compared to repository time frame, experimental observations. The highest-level model is purely mechanistic and is based on first principles. When purely mechanistic models are unavailable or unobtainable, semi-empirical models can be utilized. Lastly, purely empirical models can be utilized that describe the observed material responses and dependencies on variables without reference to specific mechanisms. Since detailed mechanistic models or the data to support them are not available, this latter approach is utilized herein to develop the model that describes the chemical conditions that cause pitting of zirconium cladding (Section 6.2.5).

6.2.1 Cladding Condition As-Received

The important element of the cladding condition as-received at the repository is the fraction of failed cladding, which is taken from the analysis by S. Cohen & Associates (1999 [DIRS 135910], p. 6-11, Table 6.2, Row 1). Incipient failures were not included in that analysis. This reference gives an expected value of as-received failed cladding as 0.1% with a range of 0.01% to 1%. The expected value is interpreted to be the median value. A log uniform distribution is applied to the range 0.01 to 1 because it gives equal weight to the full distribution and addresses the large (two orders of magnitude) size of the uncertainty. The median value of 0.095% and a maximum value of 1.29% calculated in *Clad Degradation – Summary and Abstraction* (CRWMS M&O 2001 [DIRS 151662], p. 65) corroborates the median and maximum values obtained by S. Cohen & Associates (1999 [DIRS 135910]). The uncertainties in terms of the range are carried forward into the abstraction.

S. Cohen & Associates (1999 [DIRS 135910]) reviewed the cladding conditions and failure modes for as-received fuels. Table 6-1 gives their components of failure percents and causes. S. Cohen & Associates (1999 [DIRS 135910], p. 2-31) also concluded that damage of fuel during shipment, if any, would be minor. The as-received failure rates in this model is compared to rates from other alternative conceptual models in Section 6.3.1 and to other reported values in Section 7.4.1.

Table 6-1. Fuel Failure Rates

Fuel Service Period	Rod Failure %
In-Service	< 0.05
Pool Storage	0
Dry Storage	0.03
Consolidation	0.005
Other Handling	0.0003
Total	< 0.1

Source: S. Cohen & Associates 1999 [DIRS 135910], p. 7-1.

The as-received failed fuel rods are available to undergo clad splitting and fuel-pellet corrosion when the waste package fails. This model parameter is summarized for TSPA-LA input as Item 1 of Table 8-1.

6.2.2 Stainless Steel Cladding Distribution

The stainless steel cladding degradation model uses both the fraction of stainless-steel-clad fuel assemblies in the repository and the distribution of these assemblies among the spent fuel-containing waste packages.

The total number of fuel rods with stainless steel cladding is fixed since there are no operating reactors using this cladding. As indicated by S. Cohen & Associates (1999 [DIRS 135910]), there are 2,179 such assemblies in storage, either at the reactor sites or at General Electric's Morris, Illinois facility. This total consists of 1,846 PWR assemblies (84.7% of the total) and 333 BWR assemblies (15.3% of the total). If these are loaded into 21-PWR and 44-BWR waste packages containing only stainless steel cladding, this represents 88 PWR waste packages (1,846 PWR assemblies/21 PWR assemblies per waste package) and 8 BWR waste packages (333 BWR assemblies/44 BWR assemblies per waste package). However, it is likely that the approximately 10% of the fuel that is grossly failed (EPRI 1996 [DIRS 160968]) would not be included in the mandated 60,000 MTU of SNF (grossly failed fuel would likely be relegated to the end of the line of shipments since they represent only a small fraction of the total and the requirements for such shipments have not yet been developed). Thus, the total of 96 waste packages would be reduced to about 86 waste packages that would be included in the repository loading of 7,472 CSNF waste packages given in Table 4-2 of *Initial Radionuclide Inventories* (BSC 2004 [DIRS 170022]). This then represents about 1.15% of the total number of waste packages containing commercial SNF. As noted in Assumption 5.1, each receipt of stainless-steel-clad fuel could be loaded into three waste packages, bringing the percent of waste packages that could contain stainless steel up to about 3.5%.

The 7,472 waste packages contain 220,810 CSNF assemblies (BSC 2004 [DIRS 170022], Table 4-2). Since there are 2,179 assemblies with stainless steel cladding, the percent of assemblies in the repository with stainless steel cladding would be 2,179×100/220,810 or 0.99% (rounded to 1.0%). Since it is assumed that 3.5% of the waste packages contain stainless-steel-clad assemblies (Assumption 5.1), then on average a waste package with stainless-steel-clad assemblies would contain 1.0%×100/3.5% or 28.6 % stainless steel and 71.4% Zircaloy-clad assemblies.

Fuel rods with stainless steel cladding were used in the early generation LWR core designs but are no longer used. In the TSPA-SR and the TSPA-LA models, the stainless-steel-clad fuel rods are modeled as perforated upon placement, making them immediately available for splitting when the waste package fails (See Assumption 5.3). Due to the conservative nature of the assumption, no uncertainty has been assigned to the failure probability for stainless-steel-clad fuel.

For the TSPA-LA model, an uncertainty for the distribution of stainless-steel-clad fuel into waste packages is developed here using Assumption 5.1. If the deliveries to the repository were always timed so that all deliveries always required partial loading into a new waste package, then the number of waste packages containing stainless-steel-clad fuel assemblies could double to about

7% and the stainless-steel-clad fuel rod content of these waste packages would halve to about 15%. This uncertainty is an epistemic uncertainty since it is due to the lack of knowledge of the loading of the stainless-steel-clad assemblies. Because of the general uncertainty in waste package loading patterns, the fraction of waste packages containing stainless-steel-clad fuel rods will be taken as uniformly distributed between 3.5% and 7% of the waste packages.

The percentage of stainless-steel-clad rods is calculated as follows:

$$X = \frac{100\% \times 1.0\%}{Y} \qquad \text{(Eq. 6-1)}$$

where

X = percentage of stainless-steel-clad rods in the waste package
Y = percentage of waste packages containing stainless-steel-clad fuel rods.

The parameters for the TSPA-LA model are summarized as Items 2 and 3 of Table 8-1. No corroboration of this model is necessary since the approach is reasonable. It is also noted that the post-closure performance of the repository as developed in the TSPA-LA is not expected to be sensitive to the distribution of the stainless-steel-clad assemblies within waste packages.

6.2.3 Mechanical Failure of Cladding

The two forms of mechanical failure of the cladding considered in the TSPA-LA model are seismic-induced failure (including rockfall) and mechanical failure from the static load of a rock overburden. Rockfall will not directly impact the cladding since the waste package and drip shield will protect the waste form until they fail by corrosion processes (See Assumption 5.2). Thus, rock overburden is the primary failure mechanism.

The definition of the mathematical relationship for cladding failed due to rock overburden is in two steps:

i. Cladding failure will begin when both the fraction of drip shield patch openings and the fraction of waste package patch openings exceed the sampled uncertainty (0.2 to 0.5, uniform distribution)

ii. After failure begins, the cladding failure fraction will depend linearly on only the fraction of waste package patches open (not the sum of waste package and drip shield patch fractions).

In the unlikely event that both the waste package and drip shield fail, a rubble bed of rocks from the collapse of the drift is assumed to mechanically load the exposed fuel rods. This load will fail the cladding (CRWMS M&O 1999 [DIRS 136105], Section 6.2). For estimating the peak dose, rubble bed damage to the cladding can not be neglected. The proposed model uncertainty has been assumed (Assumption 5.2). Failure of the cladding starts when a minimum of 20%, with a uniform distribution of 20% to 50%, of the patches on both the drip shield and waste package surface are corroded open. The fraction of rods failed is then modeled to increase linearly with increased waste package surface opening (patches) until all the rods are failed when 50% of the waste package patches are open (Assumption 5.2). The TSPA-LA model contains a submodel (BSC 2004 [DIRS 168504], Section 6.3.5) that predicts when patches (defined areas of the waste package surface) are corroded through. This is an epistemic uncertainty since it is due to a lack of knowledge of how the waste package and drip shield will fail and when the weight of the rocks will be loaded onto the fuel. The cladding will fail using the following relationship (Assumption 5.2):

$$\% \text{ Rods failed} = 200 \times (PF - X) \qquad \text{(Eq. 6-2)}$$

where:

PF = waste package patch fraction, in range $X \leq PF \leq X+0.5$
X = random number, uniformly distributed in range $0.2 \leq X \leq 0.5$

The percent of rods failed is limited to less than or equal to 100%. This model parameter is summarized for TSPA-LA input as Item 4 of Table 8-1. No corroboration of this model is necessary since the approach is conservative.

6.2.4 Split Cladding

After the waste package and cladding fail, water or moist air can enter the fuel rod and interact with the fuel pellets. This can cause an increase in the volume of the pellet due to the formation of higher oxides of uranium (such as schoepite). This atmosphere can also lead to oxidation of the exposed inner surface of the cladding. The increase in volume of the pellet or cladding interior surface will cause axial splitting of the cladding. Over time, this splitting will extend over the full active length of the rod (listed as 366-cm in Item 6 of Table 4-1), exposing a greater fuel surface area to the environment. Cladding with pre-existing flaws splits when the waste package fails (Assumption 5.3). Intact cladding starts to split when the cladding fails but only after waste package failure. ANL has performed fuel degradation tests with two intentionally perforated fuel segments in humid air at 175°C (Cunnane et al 2003 [DIRS 162406]). These conditions are more severe than the anticipated repository conditions, since the cladding will approach ambient temperatures after a few thousand years. For early waste package failure at 500 years, the waste package surface temperatures are in the range of 105°C to 120°C (BSC 2004 [DIRS 170019], Figure A-6). After waste package failure, the waste package interior atmosphere will become air saturated with humidity but not pure steam. Both of the ANL fuel segments split in less than two years, a rapid time scale in terms of the TSPA model. As a result, in the TSPA-LA model, the cladding is assumed to instantly split, within one TSPA time step (Assumption 5.3), leaving the fuel pellets exposed for corrosion. No uncertainty is assigned to the model since it is simple and results in splitting being effectively instantaneous over TSPA-LA repository time steps.

After the cladding has split, it still surrounds the fuel pellets, holds them into the original fuel geometry, and limits the amount of water or moist air that contacts the fuel and limits the release of radionuclides. However, in this model, the entire surface area is conservatively assumed to be available for corrosion processes to occur. While rapid axial splitting of fuel rod cladding has been observed in the ANL tests and in operating BWR reactors, such splitting has not observed in spent fuel pools for time periods of 30 years. Neither has axial splitting been observed in operating PWR reactors where there is less coolant boiling, a more reducing environment due to a hydrogen overpressure, and thus a lower concentration of dissolved oxygen in the water. Only the UO_2 in failed fuel rods is available to corrode. After splitting, the fuel pellets along the complete length of the rod are available to corrode. Corrosion starts when the waste package fails and water or moist air enters. The width of the split increases with time as the UO_2 corrodes to form a secondary alteration product, commonly called the rind.

Table 6-2a and Table 6-2b describe the mathematical relationships for how the split opens as the fuel pellet corrodes. For fuel under repository conditions, the rind is schoepite. Schoepite (UO_3:$2H_2O$) is formed when the fuel (UO_2) reacts with either water or moist air at temperatures below 90°C. At higher temperatures, dehydrated schoepite forms. Given the short time that higher temperatures exist after waste package breach, using schoepite properties is a good approximation and is conservative since schoepite has a lower density than dehydrated schoepite. The mass of the rind is increased from the UO_2 mass by the ratio of the molecular weights of the two minerals since water and oxygen are absorbed to form schoepite. The density of the (schoepite) rind, including porosity, ranges from 3.4 to 4.6 g/cm^3 with a mean value of 4.0 g/cm^3 (combination of Items 1 and 2 from Table 4-1). The uncertainty in the porosity input is carried forward through the model to be uncertainty in diffusion parameters. The small amount of porosity of the unreacted UO_2 fuel pellets is neglected in the density value given in Table 6-2a and Table 6-2b.

The volume ratio (volume rind/volume UO_2) is calculated using Equation 6-3, shown in Table 6-2b, and ranges from 2.8 to 3.8 with a mean value of 3.3. Note that the small amount of porosity in the unreacted UO_2 is conservatively ignored. The total volume of fuel in the waste package is calculated with Equation 6-4 using Items 5 through 7 in Table 4-1 and recognizing that there are 5,544 rods (21 assemblies $\times$ 264 rods per assembly) in the 21-PWR assembly waste package. Both the corrosion rate and the amount of fuel that is corroded are calculated by the TSPA code using the temperature and chemistry conditions. The number of failed rods is the product of the total number of rods times the rod failure fraction (Equation 6-5). The volume of unreacted fuel in a rod is given in Equation 6-6. The volume of the rind in a fuel rod is the product of the volume of corroded UO_2 times the volume rind multiplier (Equation 6-7). For conservatism, the small amount of uranium that dissolves is ignored and all the uranium is taken to be precipitated as rind. The cladding is filled with UO_2 and rind giving a new diameter based on the equation for the volume of a cylinder (Volume = $(\pi/4)D^2$). Equation 6-8 gives the new rod diameter.

Table 6-2a. Input Variables to Calculate Cladding Split Geometry

Name	Units	Description	Definition
WW-UO_2	g/mol	Molecular weight of UO_2	270 g/mol
DUO_2	g/cm^3	Density of UO_2	10.97 g/cm^3
Dsch	g/cm^3	Theoretical density of schoepite	4.83 g/cm^3
Por	None	Porosity in schoepite	0.05 to 0.30, uniformly distributed
Lr	cm	Rod length	3.66 m
Nr	None	Number of Rods per waste package	5544 (264x21 assemblies/WP)
Dinit	cm	Initial pellet diameter	0.819 cm
Frod(t)	None	Fraction of rods that are failed	0.0001-0.01, mean of 0.001
Fcor(t)	None	Fraction of total waste package inventory that has corroded	WAPDEG calculation (not calculated in this AMR)

Source: Table 4-1

Table 6-2b. Calculated Variables and Equations to Calculate Cladding Split Geometry

Name	Units	Description	Definition	Equation
VRM	None	Volume rind ratio multiplier	(MW-sch/MW-UO2)*DUO$_2$ / ((1 – Por) × Dsch)	6-3
VI	cm^3	Initial volume of fuel in a waste package	Nr × Lr × π × Dinit^2 / 4	6-4
Nf(t)	None	Number of failed rods	Frod(t) × Nr	6-5
Vfr(t)	cm^3	Volume of UO$_2$ in 1 failed rod available for corrosion	VI × (Frod(t) – Fcor(t)) / Nf(t)	6-6
Vrr(t)	cm^3	Volume Rind in a rod	VI × Fcor(t) × VRM / Nf(t)	6-7
Drf(t),	Cm	Diameter of rod (UO$_2$ + rind)	√ (4 × (Vfr(t) + Vrr(t)) / (Lr × π))	6-8
Dif Dist(t)	Cm	Average Diffusion distance; (volcanic case)	Drf(t) / 2; (Dif Dist(t) = Drf(t) / 4)	6-9
SO(t)	Cm	Split Opening; (volcanic case)	π × (Drf(t) – Dinit); (SO(t) = π × Drf(t))	6-10
A Dif(t)	cm^2	Area for diffusion from split	Nf(t) × SO(t) × Lr	6-11
Vw(t)	cm^3	Volume of water in the rind	Por × Vrr(t) × Nf(t)	6-12

The average diffusion distance for radionuclides from the rod to adjacent metallic corrosion products in the waste package is half of the rod diameter (Equation 6-9). For the volcanic scenario, the cladding is conservatively assumed to be oxidized and provides no barrier to the release of radionuclides. Thus, for this case, the average diffusion distance is one fourth of the rind and the unreacted UO_2 diameter (Equation 6-9). Seismic effects are discussed in *Seismic Consequence Abstraction* (BSC 2004 [DIRS 169183]).

The split opening is the difference between the current rod circumference and the initial rod circumference (Equation 6-10). The split can occur at any orientation around the rod circumference. Since, the conversion of UO_2 to schoepite does not lead to additional cracking or powdering, relocation is not considered in the model. The total area for diffusion of radionuclides from the rind in the failed rods to adjacent metallic corrosion products in the waste package is the product of rod length, split opening, and number of failed rods in a waste package (Equation 6-11). Water vapor may also collect in the pores of the rind. The volume of water available for dissolution of radionuclides in the rind is equal to the porosity of the rind times the rind volume (Equation 6-12).

The split will continue to widen until all of the UO_2 in the fuel rod has corroded and then a steady-state geometry exists. The time to completely corrode the fuel rod depends on the waste package temperature and in-package chemistry but generally requires less than 2,000 years following exposure based upon the forward reaction rate for UO_2. The diffusion through the cladding split continues until all of the soluble radionuclides are released. The clad splitting parameters for the TSPA-LA model are summarized at Items 5 through 10 in Table 8-1.

6.2.5 Localized Corrosion Degradation

Pitting of zirconium alloy cladding was discussed in *Pitting Model for Zirconium-Alloyed Cladding* (BSC 2004 [DIRS 170043]). In that report, the potential for crevice corrosion and stress corrosion cracking was also briefly examined. Crevice corrosion was ruled out based on the lack of observations in the literature and in DOE experiments. Stress corrosion cracking (SCC) was also found to be unlikely since SCC would likely occur under the same environmental conditions as pitting and pitting was ruled out as discussed below. Justification of the decision to exclude from the TSPA-LA FEPs 2.1.02.15.0A, 2.1.02.16.0A, and 2.1.02.21.0A, related to localized corrosion of the cladding (see Table 6-1) is documented in *Clad Degradation – FEPs Screening Arguments* (BSC 2004 [DIRS 170019]).

Zirconium alloys are susceptible to pitting in a particularly aggressive combination of chloride (Cl^-) ions, ferric ions (Fe^{+3}) or hydrogen peroxide (H_2O_2). In order to predict cladding failure from chloride pitting, a review of the literature for pitting rates and electrochemical data for various zirconium alloys was conducted (see BSC 2004 [DIRS 170043]). Based on this review of the literature, failure criteria were constructed based on an electrochemical definition of pitting as the condition at which the corrosion potential exceeds repassivation potential (i.e., $E_{corr} > E_{rp}$). Electrochemical values were obtained for zirconium alloys in various solution concentrations of Cl^-, Fe^{+3}, and H_2O_2 using data obtained from various experimental sources. The model to predict repassivation potential depends only on chloride concentration in the solution. The corrosion potential (E_{corr}) was modeled by doing a regression analysis to fit experimental data with varying molar concentrations of Cl^-, Fe^{+3}, and hydrogen peroxide (H_2O_2). The range of concentrations used in the regression analysis is higher than the conditions expected in the repository. The model predicts the conditions where experimental pitting was observed. In-package chemistry analysis (that included radiolysis) was performed. When in-package chemistry results were combined with the cladding pitting model, no cladding failures were predicted and these FEPs (2.1.02.15.0A and 2.1.02.16.0A, see Table 6-1 in BSC 2004 [DIRS 170019]) were excluded from the TSPA-LA model.

6.3 ALTERNATIVE CONCEPTUAL MODELS

The TSPA performed for the repository in 1995 (CRWMS M&O 1995 [DIRS 100198]) did not consider the effects of cladding. They consider the fuel as bare pellets. Bare pellets can be considered an alternative conceptual model that is extremely conservative because failure to consider cladding permits all fuel to dissolve at the intrinsic dissolution rate. A solubility limit, representing the solubility of the secondary phase, is used to limit the release rate of the fuel. In all TSPA calculations that include cladding degradation, one of the sensitivity studies is always the effect of removing the cladding. These sensitivity studies could also be considered alternative conceptual studies. Sensitivity studies (BSC 2003 [DIRS 168796], p. 3-13 and Table 2) compared TSPA with and without cladding and showed that the cladding model only had a small effect on the TSPA results and therefore was given an importance to expected risk as "Not Significant."

Analyses performed in support of possible repository sites in Europe also did not consider cladding as a barrier to radionuclide release. Most European reactors are run at higher temperatures and go to higher burnup than U.S. reactors. The potential European sites are saturated, reducing, and hydrogen over-pressured environments where the UO_2 corrosion rates are slow. As a result, in-repository cladding degradation would be slow, but the potential for damage prior to placement would be greater. Thus, cladding credit was not considered.

The cladding degradation model has evolved, and the earlier repository studies could also be considered alternative conceptual models. *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998

[DIRS 100362]) was the first study to include cladding as a barrier to radionuclide release. Cladding was included in *FY01 Supplemental Science and Performance Analyses, Volume 2: Performance Analyses* (BSC 2001 [DIRS 154659]) and *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000 [DIRS 153246]). The Electric Power Research Institute (EPRI) included a cladding degradation model in their recent analysis (EPRI 2000 [DIRS 154149], p. 5-35) based on general corrosion. Their model concludes that cladding reduces doses in the periods 30,000 to 200,000 years (EPRI 2000 [DIRS 154149], p. 13-5).

Siegmann and Devonec (2002 [DIRS 160787]) performed a series of sensitivity studies that showed the effect of using many of these alternative conceptual models in *Total System Performance Assessment for the Site Recommendation* (TSPA-SR) (CRWMS M&O 2000 [DIRS 153246]). This study concluded that the exclusion of the cladding degradation model would increase the peak dose rate by only about a factor of three from the TSPA-SR model.

Thus, this alternative conceptual model was rejected since it is overly conservative. As noted above, this condition will be evaluated via a sensitivity analysis.

6.3.1 Cladding Condition As-Received

In the TSPA-SR model, the values used for the fraction of cladding that has failed prior to emplacement are an alternative conceptual model (CRWMS M&O 2001 [DIRS 151662], p. 65). This analysis developed values (0.095% median value, range 0.0155% to 1.29%) that are close to the values developed by S. Cohen & Associates (1999 [DIRS 135910], Table 6.2) to be used in the TSPA-LA model (0.1% median value, range 0.01% to 1%). Since they are similar, the use of this alternative conceptual model would not affect the TSPA-LA results. Another alternative concept would be to model all cladding as failed upon receipt. Siegmann and Devonec (2002 [DIRS 160787], Figure 2) show that even under this extremely unlikely scenario, the effect would only be approximately triple the peak dose compared to the base case model with no effect during the first 10,000 years.

Thus, this alternative conceptual model was rejected since it was similar to the results of the model developed in this report.

6.3.2 Stainless Steel Cladding Distribution

The quantity of commercial spent fuel with stainless steel cladding is fixed since no current fuel manufacturer is using stainless steel. An alternative conceptual model could have the stainless-steel-clad fuel distributed equally into all the waste packages rather than the base case of loading into a limited number of waste packages. This distribution could reduce the peak dose by spreading out the release from the stainless-steel-clad fuel rods. However, this is an unlikely alternative conceptual model since the current surface design calls for loading waste packages as the fuel arrives on the site and blending the stainless-steel-clad fuel is not planned. As noted in the previous paragraph, the TSPA-LA is not expected to be that sensitive to the fraction of failed fuel. Thus, this alternative conceptual model is rejected as being unlikely and nonconservative.

6.3.3 Mechanical Failure of Cladding

There are no alternative conceptual models that have been identified for the mechanical failure of the cladding.

6.3.4 Split Cladding

An alternative conceptual model to the instant splitting model that is described in this report is the dry oxidation model. The dry oxidation model (CRWMS M&O 2000 [DIRS 149230]) considers the further oxidation of UO_2 to U_3O_8. This occurs in a relatively dry environment (humidity below 50%). After an initial incubation time, the rods are observed to split very quickly as a result of the increase in volume of the U_3O_8 from UO_2. Because of humidity and temperature limits, it is unlikely that dry splitting will occur in a postclosure waste package. The instant splitting model presented in this model report is conservative with respect to splitting times if dry oxidation should occur. Thus, the alternative dry splitting model is rejected since it is unlikely and leads to results similar to the wet splitting model.

The rind model used for the TSPA-SR model could be considered an alternative conceptual model for the release of radionuclides from the split rod. The rind volume was just the original volume of the UO_2, increased by porosity. The presence of the split cladding was neglected and the only diffusion considered was through the crack or patch in the waste package. Since this approach leads to very conservative releases, this alternative conceptual model was rejected.

The alternative conceptual models considered for cladding degradation are summarized in Table 6-3. The summary shows (Column 3 of Table 6-3) that most of the alternative conceptual models are less appropriate for use in the TSPA-LA model because they are overly conservative or reflect conditions not expected in the repository.

Table 6-3. Alternative Conceptual Models Considered

Alternative Conceptual Model	Key Assumptions	Screening Assessment and Bases
No Cladding Credit	Bare fuel when waste package fails.	Overly conservative. Addressed in TSPA-LA as a sensitivity study.
TSPA-SR Initial Cladding Failures	Addresses various causes of fuel failure.	Similar results to this model.
Stainless-Steel-Clad Fuel Evenly Distributed	Stainless-steel-clad fuel evenly distributed in all waste packages.	Even distribution considered highly improbable. Not conservative.
Splitting – Dry Oxidation	UO_2 forms U_3O_8.	Unlikely because it requires low humidity, high temperatures, produces instant splitting as current model.
TSPA-SR Rind Model	No cladding present, only diffusion out of waste package.	Overly conservative. Similar to No Cladding Credit.

6.4 FEATURES, EVENTS, AND PROCESSES ADDRESSED

The development of a comprehensive list of FEPs potentially relevant to postclosure performance of a repository at Yucca Mountain is an ongoing, iterative process based on site-specific information, design, and regulations. Particular FEPS are either included or excluded depending on their relevance to each model report in accordance with their assignment in the LA FEP list (DTN: MO0407SEPFEPLA.000 [DIRS 170760]).

Table 6-4 provides a list of FEPs that are used in this model analysis and in the TSPA-LA. Specific reference to the various sections within this document where issues related to each FEP are addressed is provided in the table. The detailed discussion of these FEPs, implementation in TSPA-LA and the exclusionary arguments are documented in *Clad Degradation – FEPs Screening Arguments* (BSC 2004 [DIRS 170019]).

Table 6-4. Included FEPs for this Model Report

FEP Number	LA-FEP Name	Section Where Disposition is Described
LA:2.1.02.12.0A	Degradation of cladding prior to disposal	Sections 6.2.1, 6.2.2
LA:2.1.02.23.0A	Cladding Unzipping (Axial Splitting)	Sections 6.2.4, 6.3.4
LA:2.1.09.03.0A	Volume increase of corrosion products impacts cladding	Sections 6.2.4, 6.3.4
LA:2.1.02.25.0B	Naval SNF cladding	Section 6.0

6.5 MODEL UNCERTAINTIES

Uncertainties from both input and model development were carried forward to the model abstraction. The input values for the fraction of cladding failed as-received contained two orders of magnitude uncertainty. The fraction of CSNF that is clad in stainless steel is fixed and no uncertainty was considered. The input loading pattern for the stainless-steel-clad fuel into waste packages did not have any uncertainty but one was developed in this model report. No uncertainty is used for the fraction of stainless-steel-clad fuel that is failed; an upper limit is used. There are no direct input uncertainties to the mechanical failure submodel that need to be propagated to the abstraction. An uncertainty in when the cladding starts to fail is introduced in this model report and carried forward to the abstraction. A conservative model of instant splitting of the cladding was used without any uncertainty. The uncertainty in rind porosity was included in the input and is carried forward to the model abstraction. There is also an uncertainty in UO_2 corrosion rates and chemical and temperature environments inside the waste package. These uncertainties are generated in the TSPA-LA calculations and are carried forward into the abstraction and produce an uncertainty in the rind geometry and diffusion of radionuclides from the fuel rind. The uncertainties in the cladding degradation model are epistemic type.

7. VALIDATION

This section contains a discussion of purpose (Section 7.1), level of confidence required (Section 7.2), criteria used (Section 7.3), and a discussion of the activities performed to generate confidence in the model. The discussion below summarizes the effort performed to develop confidence during model development. The activities performed to generate confidence in the model after model development are given in Section 7.4.

Confidence-Building During Model Development—Section 5.3.2(b) of AP-SIII.10Q defines activities required to ensure that validation of the mathematical model and its underlying conceptual model includes documentation of decisions or activities implemented to generate confidence in the model during model development. These requirements are augmented by Section 2.2.1 of *Technical Work Plan for: Regulatory Integration Modeling and Analysis of the Waste Form and Waste Package* (BSC 2004 [DIRS 171583]) as well as AP-2.27Q. The development of the model addressed in this report was conducted according to these criteria. The appropriate references to AP-SIII.10Q and AP-2.27Q are provided after each activity, as follows:

> 1. *Selection of input parameters and/or input data, and a discussion of how the selection process builds confidence in the model*. (AP-SIII.10Q, 5.3.2(b)(1); AP-2.27Q, Attachment 3, Level I (a))

The bases for selecting the input data that are used to determine and develop the model are documented Section 4.1. Discussion regarding the selection and appropriateness of the data is documented in Section 4.1. Model assumptions have been described in Section 5. Detailed discussion about model concepts can be found in Sections 6.2.1 to 6.2.4. Thus, this requirement is considered to be satisfied.

> 2. *Description of calibration activities, and/or initial boundary condition runs, and/or run convergences, simulation conditions set up to span the range of intended use and avoid inconsistent outputs, and a discussion of how the activity or activities build confidence in the model. Inclusion of a discussion of impacts of any non-convergence runs* (AP-SIII.10Q, 5.3.2(b)(2); AP-2.27Q, Attachment 3, Level I (e)).

Section 6 describes the approach to model development. Sections 6.2.1 through 6.2.4 discuss the various parts of the model along with the corroborating data that add confidence in the model. These sections describe the range of experimental conditions under which the data were collected and also compare these conditions to those anticipated in-package conditions. Section 6.2.5 contains a discussion of other corrosion modes that could impact performance of the zirconium cladding. The models are basically either conservative assumptions on cladding condition or utilization of the available data; and, hence run convergences or nonconvergences are not applicable. Thus, this requirement is considered to be satisfied.

> 3. *Discussion of the impacts of uncertainties to the model results including how the model results represent the range of possible outcomes consistent with important uncertainties.* (AP-SIII.10Q, 5.3.2(b)(3); AP-2.27Q, Attachment 3, Level 1 (d) and (f)).

Uncertainties associated with the data used to determine the model's parameter values and the uncertainties in the abstracted model are discussed in Sections 6.2 and Section 6.5, respectively.

> *4. Formulation of defensible assumptions and simplifications*. (AP-2.27Q, Attachment 3, Level I (b)).

Discussion of assumptions and simplifications and their rationale are provided in Section 5 and Sections 6.2, respectively.

> *5. Consistency with physical principles, such as conservation of mass, energy, and momentum*. (AP-2.27Q, Attachment 3, Level I (c))

The empirical approach to model development is described in Section 6. An empirical approach is used, since mechanistic models of pitting corrosion or knowledge of the complete set of the underlying physical principles were not readily available. This is consistent with industry practice. The development of the model is described in Section 6.2. The model utilizes the results of experiments conducted over a broad range of expected environments.

The following list (in approximate order of citation) gives the data, models, information, and sources used to complete the validation activities:

1. *Risk Information to Support Prioritization of Performance Assessment Models* (BSC 2003 [DIRS 168796]), sensitivity studies

2. "Sensitivity Studies of the Effect of Cladding Degradation on TSPA Results" (Siegmann and Devonec 2002 [DIRS 160787]), sensitivity studies

3. *An International Peer Review of the Yucca Mountain Project TSPA-SR, Total System Performance Assessment for the Site Recommendation* (OECD and IAEA 2002 [DIRS 158098]), peer review

4. *Clad Degradation – Summary and Abstraction* (CRWMS M&O 2001 [DIRS 151662]), as-received fuel failures), and causes of fuel failures

5. *Dry Cask Storage Characterization Project* (EPRI 2002 [DIRS 161421]), dry storage tests, testing of fuel from dry storage canisters

6. "Transportation in France" (Debes 1999 [DIRS 161193]), fuel failures during transport

7. Assorted references listed in Table 7-1, fuel reliability

8. *Effectiveness of Fuel Rod Cladding as an Engineered Barrier in the Yucca Mountain Repository* (S. Cohen & Associates 1999 [DIRS 135910]), reliability of stainless steel-clad fuel

10. *Breakage of Commercial Spent Nuclear Fuel Cladding by Mechanical Loading* (CRWMS M&O 1999 [DIRS 136105]), breakage of fuel from mechanical loads

11. *Transportation Accident Scenarios for Commercial Spent Fuel* (Wilmot 1981 [DIRS 104724]), structural strength of spent nuclear fuel

12. *Shipping Container Response to Severe Highway and Railway Accident Conditions* (Fischer et al. 1987 [DIRS 104774]), structural strength of spent nuclear fuel

13. "Dynamic Impact Effects on Spent Fuel Assemblies" (Witte et al. 1989 [DIRS 102158]), structural strength of spent nuclear fuel, failure rates of fuel rods

14. *A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements* (Sanders et al. 1992 [DIRS 102072]), structural strength of spent nuclear fuel

15. "A Review of Fuel Degradation in BWRs" (Edsinger 2000 [DIRS 154433]), splitting of cladding

16. "Axial Splits in Failed BWR Fuel Rods" (Lysell et al. 2000 [DIRS 154432]), splitting of cladding

17. *Results from NNWSI Series 1 Spent Fuel Leach Tests* (Wilson 1985 [DIRS 102147]), *Results from Cycles 1 and 2 of NNWSI Series 2 Spent Fuel Dissolution Tests* (Wilson 1987 [DIRS 102150]), *Results from NNWSI Series 3 Spent Fuel Dissolution Tests* (Wilson 1990 [DIRS 100793]), release of radionuclides through split cladding.

7.1 INTENDED PURPOSE OF THE MODEL

The purpose of the commercial spent nuclear fuel (CSNF) cladding degradation model is to develop the summary cladding abstraction that will be used in the Total System Performance Assessment for the License Application (TSPA-LA). The model will describe how the cladding fails and what fraction of the fuel matrix, contained inside the cladding, is exposed to the waste package environment. The cladding degradation model consists of four submodels, cladding condition as-received, stainless steel cladding, mechanical failure of cladding, and clad splitting. This model will be used in the TSPA-LA model to define what fraction of the fuel matrix is available for dissolution (corrosion) as a function of time and the TSPA-LA parameters are described in Table 8-1. Repository data used in this model is identified in Table 4-1 and validation of the model is presented in the specific submodel discussions that follow.

7.2 DETERMINATION OF THE LEVEL OF CONFIDENCE REQUIRED

The cladding degradation summary and abstraction model is used to describe how the cladding degrades in the waste package during the postclosure period. *Technical Work Plan for: Regulatory Integration Modeling and Analysis of the Waste Form and Waste Package* (BSC 2004 [DIRS 171583], Section 2.2.2) concluded that the degradation rate of the cladding required the lowest confidence level (Level I) for validation. This was because sensitivity studies (BSC 2003 [DIRS 168796], Section 3.3.6 and Table 2) showed that the cladding model had only a small influence on the TSPA-SR model results and, therefore, was given an importance to expected risk as "Not Significant." In a peer-reviewed paper, Siegmann and Devonec (2002 [DIRS 160787], Figure 2) also show that the TSPA results for peak dose are only weakly sensitive to the cladding model. Therefore, the lowest level of confidence is deemed sufficient to validate this model. Postdevelopment model validation of the cladding degradation model is accomplished by corroborating model predictions with results of alternative mathematical models and/or experimental data developed independently and available in the published literature and industry reports. Validation is performed independently for each part of the cladding degradation model and is discussed below.

7.3 CRITERIA USED TO DETERMINE THAT THE REQUIRED LEVEL OF CONFIDENCE HAS BEEN OBTAINED

The technical work plan for this model report (BSC 2004 [DIRS 171583], Table 2-1) lists the four activities, stated as criteria, that have been selected to validate this model report. Corroboration has been achieved when data or model results used for validation, obtained under similar environmental conditions, match qualitatively and are bounded by the uncertainty band of the model.

> *Criterion One: Is the waste form degradation model consistent with the experimental data generated for the repository at various laboratories? When the model is not consistent, are there physical or phenomenological reasons why the differences are observed?*
>
> *Criterion Two: Is the database of waste form degradation rates in the model consistent with the rates published in the peer-reviewed or industrial literature?*
>
> *Criterion Three: Has the model been corroborated with results from an alternative mathematical model?*
>
> *Criterion Four: Has the model been corroborated by a peer review per AP-2.12Q, Peer Review, or been published in a refereed professional journal?*

7.4 ACTIVITIES PERFORMED TO GENERATE CONFIDENCE IN THE MODEL

Postdevelopment model validation of this model was accomplished by corroborating model predictions with results of alternative mathematical models, experimental results developed independently and available in the published literature and industry reports, or through the use of peer review.

7.4.1. Cladding Condition As-Received

Criteria 2 and 3 are used to validate the Cladding Condition As-Received submodel. The initial failure percentage for the rods in a waste package in the TSPA-LA model is represented by a log uniform distribution with a range 0.01% to 1.0%. This distribution produces a median value of 0.1%. It was developed by S. Cohen & Associates (1999 [DIRS 135910], Table 6.2). The TSPA-LA parameters for this submodel are given in Row 8 of Table 4-1. These failures include in-reactor failures, dry storage failures, and other causes (Table 6-1). Model validation of the as-received cladding failures is accomplished by validating model predictions with results of an alternative mathematical model (Criterion 3) and experimental data developed independently and available in the published literature and industry reports (Criterion 2). This also is justification of the use of technical information (as-received failure rates) by S. Cohen & Associates (1999 [DIRS 135910]). An alternative mathematical model (CRWMS M&O 2001 [DIRS 151662], p. 65), reviewed the various causes of fuel damage and independently produced very similar results (median 0.095%, range 0.0155% to 1.29%) for the rod failure rate for all causes. These results are close to the results achieved with the base model and validates this model to the required level of confidence (satisfying Criterion 3 from Section 7.3).

The initial cladding failure rate from reactor operation and associated activities is further validated using experimental data developed independently and available in the published literature and industry reports (Criterion 2). Table 7-1 gives the failure rate reported by others for various times and conditions. These support the values used in the cladding model. ANL has tested rods that have been in dry storage for approximately 15 years and have reported no signs of cladding degradation (EPRI 2002 [DIRS 161421]). Fuel failures from transportation are unlikely and not included in Table 2. Debes (1999 [DIRS 161193], p. 2) reported that no anomalies, such as fuel leakage, have been detected after transporting 27,000 assemblies in France.

Both Criteria 2 and 3 have been utilized to validate this portion of the model and, as a result, the required level of confidence has been achieved. In addition, an international peer review (Criteria 4) of the TSPA-SR model was performed that included the cladding model (OECD and IAEA 2002 [DIRS 158098], p. 33). The peer review found that the approach taken by the U.S. DOE for cladding was appropriate. They also stated that:

> The issue of cladding performance is important because it is one area of possible optimism and because it has a major effect on system performance beyond 10,000 years.

In conclusion, since the criteria have been met, no further activities are needed to complete this model validation for its intended use.

7.4.2 Stainless Steel Cladding Distribution

The model has the waste packages loaded with the stainless-steel-clad fuel as it is delivered to the site, and any delivery can require three waste packages (Assumption 5.1). Since the stainless-steel-clad fuel is modeled as failed, their placement could affect the peak dose. This assumption leads to a value of 3.5% of the waste packages that could contain stainless steel cladding and ~30% of the cladding in the content of an affected waste package that would be stainless steel. If the value doubles from 3.5% to 7% in the number of waste packages containing stainless-steel-clad fuel, the content would decrease from 30% to 15%. The fraction of stainless-steel-clad fuel in a waste package is based upon the fraction of stainless-steel-clad assemblies and loading pattern noted above. Modeling that all of the stainless-steel-clad fuel rods are failed upon placement within the waste packages is a conservative upper limit. An alternative conceptual model for stainless steel cladding in Section 6.3.2 indicated that distributing the stainless-steel clad fuel rods among all waste packages was not conservative in that it spread out the release and lowered the peak dose.

A reasonable assumption has been made regarding the distribution of stainless-steel-clad fuel rods within waste packages. Thus, no validation of this submodel is required and no further activities are needed to complete this model validation for its intended use.

Table 7-1. Comparison of Fuel Reliability from Various Sources

Fuel	Period	Reference	Failure Rate[a], %
GE-8 × 8	1983	Bailey et al. 1985, p. 1-3 [DIRS 109191]	0.007
PWR-French	1979 to 1984 1984	Dehon et al. 1985 [DIRS 109197], p. 2-24	0.001 - 0.01 0.005
BWR-Japan PWR-Japan	To 1997	Sasaki and Kuwabara 1997 [DIRS 102074], pp. 13, 14	0.01 0.002
PWR-CE	To 11/1984	Andrews and Matzie 1985 [DIRS 109190], p. 2-42, Table 2	0.011
All	Through 1984	EPRI 1997 [DIRS 100444], p. 4-1	0.02 - 0.07
All	After 1984	EPRI 1997 [DIRS 100444], p. 4-2	0.006-0.03
BWR PWR	To 1986	Sanders et al. 1992 [DIRS 102072], p. I-36	0.10 - 0.73 0.07 - 0.48
PWR-Westinghouse	1 core, debris damage after steam generator replacement	McDonald and Kaiser 1985 [DIRS 101725], pp. 2-4 and 2-5	0.26
All	1969 to 1976	Manaktala 1993 [DIRS 101719], pp. 3-2 and 3-3, Figure 3-1	0.01 - 2+
PWR-Mark B-B&W	1986 to 1996	Ravier et al. 1997 [DIRS 102068], p. 34, Figure 4	0 - 0.055
BWR	2000	Edsinger 2000 [DIRS 154433], p. 162	0.0005
BWR, GE Fuel	1995 to 1999	Potts 2000 [DIRS 160783], p. 502, Figure 1	0.00058
PWR, Mitsubishi Fuel	1992 to 1999	Doi et al. 2000 [DIRS 160781], p. 443	0 rod failures

NOTES: B&W = Babcock & Wilcox; CE = Combustion Engineering GE = General Electric; W = Westinghouse.
[a] Failure rates are on a rod basis unless noted as assembly-based. The assembly value represents the percentage of assemblies that contain at least one failed rod.

7.4.3 Mechanical Failure of Cladding

The mechanical failure model describes the fraction of cladding within the waste package that is failed as a function of the fraction of the waste package surface area open as a result of corrosion on the waste package surface. Postdevelopment model validation of the mechanical failure model is accomplished by corroborating model predictions with results of alternative mathematical models and experimental data developed independently and available in the published literature and industry reports (Criteria 2 and 3).

Failure of the cladding from the force of the rock overburden qualitatively resembles the structural analysis in Section 5.3 of *Breakage of Commercial Spent Nuclear Fuel Cladding by Mechanical Loading* (CRWMS M&O 1999 [DIRS 136105]). This analysis concluded that the fuel could not support the weight of the rock overburden. This overburden mechanically presses on the fuel after the drip shield and waste package deteriorate to the extent that the fuel no longer supports the weight of the rock overburden.

The robustness of the cladding to extreme forces has been addressed in many transportation studies, which show that significant forces are required to fail the fuel demonstrating that the model is conservative. These studies are dynamic rather than static, but they add confidence in the robustness of the fuel rods. Wilmot (1981 [DIRS 104724], Table VII) recommends the use of 71 g accelerations (where g is the gravitational constant) for the failure threshold for fuel rods experiencing side impacts. He references an experimental threshold of 122 g for SNF. Wilmot noted that, in drop tests, rods were bent with end impacts of 38 g but did not fail. He references experimental thresholds for end impacts of 234 g. Fischer et al. (1987 [DIRS 104774], Figure 8-3) suggested that 10% of the rods might fail with a 40 g end impact, and 100% might fail with a 100 g end impact. Witte et al. (1989 [DIRS 102158], Table 3) report that the acceleration needed to fail rods from side impact varies from 63 g to 211 g, depending on the fuel design. Sanders et al. (1992 [DIRS 102072], Attachment III) present detailed structural analysis of various assemblies under impacts and give (Table III-10) the probability of rod failure from 9 meter drops of transportation casks. All these references show the robustness of spent fuel rods to failure from mechanical loading. The literature data cited above shows that the rods have significant strength and validates the model that the rods break from a significant amount of rock overburden is conservative.

The TSPA-LA model contains a submodel that predicts when patches (defined areas of the waste package surface) are corroded through. It is intuitive that the waste package will offer some protection when the first patch opens and very little after the last patch opens. The present model, which assumes failure of the cladding when 20% to 50% of the patches are open, is considered reasonable. The required level of confidence has been achieved since this event occurs after both the waste package and drip shield have significantly deteriorated (>100,000 years).

Thus, the required lowest level of confidence has been achieved since at least one criterion of Section 7.3 has been met. No further activities are needed to complete this model validation for its intended use.

7.4.4 Split Cladding

The split cladding model consists of two parts: the splitting itself, and the alteration of the exposed fuel. The current split cladding model has the cladding instantaneously split when both waste package and cladding failures exist, leaving the exposed fuel pellets to react with the environment. This is based on two experiments at ANL (Cunnane et al. 2003 [DIRS 162406]) where the cladding split in less than 2 years. Postdevelopment model validation of the cladding splitting model is accomplished by corroborating model predictions with results of experimental data developed independently and available in the published literature and industry reports (Criterion 2).

Edsinger (2000 [DIRS 154433], p. 173) reports a lower bound velocity of splitting for a BWR rod of the range of 1 to 3 mm/hr for these experiments. Lysell et al. (2000 [DIRS 154432], p. 217) report axial splitting velocities of about 3 mm/hr after an incubation time. These velocities do not address incubation times but would lead to bare fuel pellets in a very short time compared to TSPA time steps (100 to 1,000 year time steps).

The rind model describes the release of radionuclides through the split opening. C. Wilson performed a series of tests measuring the release of radionuclides through a segment of fuel rod with a slit cut in it (Wilson 1985 [DIRS 102147]; Wilson 1987 [DIRS 102150]; Wilson 1990 [DIRS 100793]). The slit was cut into the cladding to model a split. He also performed the test with similar samples of fuel with the cladding removed. This permits a comparison of the reduction of the release in radionuclides caused by the split cladding. Wilson observed large reductions in the release of radionuclides. As an example, in his series three tests, cycles 1 and 2 (Wilson 1990 [DIRS 100793], Table 3.1), he observed the release of uranium was reduced by a factor of about 170.

Thus, the literature data corroborated the model utilizing Criterion 2, and the required level of confidence has been achieved. No further activities are needed to complete this model validation for its intended use.

8. CONCLUSIONS

8.1 OUTPUT TO THE TSPA-LA MODEL

Table 8-1 provides the parameters developed in this cladding degradation model that serve as inputs to TSPA-LA. The information in Table 8-1, along with the related uncertainties, are included in DTN: MO0306SPACLDDG.002. This DTN is restricted to repository designs where the cladding surface temperature is less than 350°C during post-closure. Some of the TSPA-LA input variables used in earlier analyses have been eliminated and details of their exclusion from the TSPA-LA model are documented in *Clad Degradation – FEPs Screening Arguments* (BSC 2004 [DIRS 170019]). The cladding degradation values used in the four components of the cladding degradation model are summarized below.

Table 8-1. Summary of Cladding Degradation Values to be Used in the TSPA-LA Model

Item	Input name	TSPA Parameter Name	Values
1	Distribution of failed cladding, as-received	Initial_Rod_Failures	Range 0.01 to 1%, log uniform distribution, Epistemic Uncertainty
2	Stainless-steel-clad fuel loading into waste packages	Frac_CSNF_Pkgs_SS	Waste packages containing stainless-steel-clad fuel = 3.5 to 7%, uniformly distributed, Epistemic Uncertainty
3	Percent of total commercial SNF inventory with stainless steel cladding	Inven_SS	1.0%
4	Rock overburden uncertainty factor	Rock_Load_Uncertainty	Uniformly distributed between 0.2 and 0.5, Epistemic Uncertainty
5	Fraction of fuel available for corrosion at any TSPA time step	Fuel_Split_Fraction	1.0
6	Density of UO2	Density_UO2	10.97 g/cm^3
7	Density of Schoepite	Density_Schoepite	Uniformly distributed between 3.4 and 4.6 g/cm^3
8	Molecular weight of UO_2	MW_UO2	270 g/mol
9	Por, Porosity in rind	Rind_Porosity_CSNF	Uniformly distributed between 0.05 and 0.3, Epistemic Uncertainty
10	Lr, Active fuel rod length, cm	Rod_Length_CSNF	366 cm
11	Nr, Number of rods per waste package	Num_Rods_WP_CSNF	5,544
12	Dinit, Initial pellet diameter	Pellet_Diameter_CSNF	0.819 cm

Output : DTN: MO0306SPACLDDG.002

8.1.1 Cladding Condition As-Received

The groups of waste packages represented in the TSPA-LA model have an initial percent of failed rods defined by the log uniform distribution with a range 0.01% to 1%. This fuel rod failure rate is based on historical data on reactor operation and includes uncertainty. It also includes failure from wet and dry storage at the reactor site, and handling. This percentage of rods is available for radionuclide release through fast release and splitting when the waste package fails. As-received cladding failures are discussed in Section 6.2.1.

8.1.2 Stainless Steel Cladding Distribution

The abstraction places the fuel with stainless steel cladding into waste packages as it arrives at the repository. This model results in 3.5% to 7% (uniformly distributed) of the waste packages contain stainless-steel-clad fuel rods. The total amount of stainless-steel-clad fuel is fixed at approximately 1.0% of the total commercial spent nuclear fuel (CSNF). These waste packages contain 30% to 15% stainless-steel-clad fuel rods, which are modeled as failed and available for instantaneous splitting when the waste package fails. Stainless-steel-clad fuel is discussed in Section 6.2.2.

8.1.3 Mechanical Failure of Cladding

Static loading from fallen rocks fails the cladding. Failure starts when both the fraction of drip shield patches and the fraction of waste package patches open exceed the sampled value for the rock overburden uncertainty fraction. This uncertainty fraction is given by a uniform distribution, sampled between 0.2 and 0.5. Failure increases linearly to 100% rod failure when 50% of the waste package patches are open. This is a reasonable and conservative model based on the very low cladding failure rates observed in the reactor industry. Mechanical failure is discussed in Section 6.2.3.

8.1.4 Split Cladding

Failed fuel rods that have previously failed are assumed to split (cladding axially splits down their length) instantaneously once the waste package has failed. This is a conservative model that bounds the results of experimental observations. It leaves the fuel pellets exposed to the waste package internal environment. For TSPA-LA purposes, this statement implies that the fraction of the fuel in a failed rod that is available for corrosion, at any time, can be set to a constant value of one. For the rind calculations, the split in the cladding slowly widens as the UO_2 corrodes because of the increase in volume of the corrosion products. The radionuclides diffuse through the split into the surrounding environment. Cladding splitting is discussed in Section 6.2.4.

8.2 CONCLUSION

In summary, the cladding degradation is analyzed in two stages in the TSPA-LA model: cladding failure and cladding splitting. The cladding degradation abstraction depends on the waste package surface patch corrosion rate. Uncertainties have been established for the important parameters, and the results vary for each TSPA-LA realization. Since the stainless-steel clad fuel, which represents about 1% of the inventory, is assumed to be failed and the Zircaloy clad fuel contains up to 1% failed rods (see Section 4.1), then less than 2% of the total as-received cladding is modeled as failed as a result of previous operations at the reactor site. The cladding is also assumed to fail from rock overburden, after significant (20% to 50%) waste package and drip shield degradation. With the ranges and uncertainties included in the abstraction, this model is valid for its intended use, analyzing cladding degradation in the TSPA-LA model. The cladding degradation model is valid for cladding peak surface temperatures below 350°C during post-closure. This is within the presently anticipated operating limits for a repository at Yucca Mountain.

8.3 YUCCA MOUNTAIN REVIEW PLAN ACCEPTANCE CRITERIA

The CSNF waste form cladding meets the definition of a barrier as stated in 10 CFR 63.2 [DIRS 156605]. The acceptance criteria for the degradation of this barrier are given below along with the manner in which this analysis report complies with those criteria. These criteria are stated in the appropriate sections of *Yucca Mountain Review Plan, Final Report* (NRC 2003 [DIRS 163274]) shown below.

8.3.1 Barriers (Section 2.2.1.1.3)

Acceptance Criterion 1 – Identification of Barriers is Adequate

> *The pertinent portions of this criterion include the identification of the barrier and its link to barrier capability.*

The description and link to the capability of cladding as a barrier is provided in Section 1 and Section 6.1.

Acceptance Criterion 2 – Description of Barrier Capability to Isolate Waste is Acceptable

> *The pertinent portions of the criterion include the capability of the barrier to reduce the release rate of radionuclides, the time over which it so functions, and the definition of the barrier.*

The capability of the barrier to reduce the release rate of radionuclides, the time frame over which it functions, and its definition are found in Section 6.1

Acceptance Criterion 3 – Technical Basis for Barrier Capability is Adequately Presented

> *The pertinent portions of the criterion include the technical basis for performance assessment and the importance of the barrier.*

The relationship of the barrier to performance assessment is described in Section 1. The importance of the barrier is discussed in Sections 6.1 and 7.1.

8.3.2 Barrier Degradation (Section 2.2.1.3.1.3)

Acceptance Criterion 1 – System Description and Model Integration are Adequate

> *The pertinent portions of this criterion include the description of the engineered barrier, the models selected or developed, the assumptions made, and the support for screening arguments for features, events, and processes (FEPs). This model is not used directly by TSPA.*

The description of cladding as a barrier is provided in Section 6.1. The models selected or developed are provided in Section 6.2. Assumptions made and their bases are found in Section 5. The FEPs addressed in this report and support for related screening arguments are found in Section 6.4.

Acceptance Criterion 2 – Data are Sufficient for Model Justification

The pertinent portions of this criterion include the parameters used to evaluate the degradation of the cladding, the available data utilized for the development of the models, and the development of the models themselves.

The parameters utilized are provided in along with the input data sources Section 4.1. The development of the models using these inputs is discussed in Section 6.2.

Acceptance Criterion 3 – Data Uncertainty is Characterized and Propagated Through the Model Abstraction

The pertinent portions of this criterion include the use of data sources within their appropriate ranges and a description of the uncertainty or variability, including probability distributions.

The data sources and their ranges of applicability are noted in Section 4.1, as well as their associated uncertainty.

Acceptance Criterion 4 – Model Uncertainty is Characterized and Propagated Through the Model Abstraction

The pertinent portions of this criterion include alternative modeling approaches and conceptual model uncertainties.

Alternative model approaches are noted in Section 6. The conceptual model uncertainties are found in Section 6.5. Alternative conceptual models are found in Section 6.3.

Acceptance Criterion 5 – Model Abstraction Output Is Supported by Objective Comparisons

The appropriate portions of this criterion include comparisons of the model output with empirical observations or other mathematical models and the margin between actual and predicted degradation of the cladding is adequate.

The validation of the models developed, using Project information, alternative mathematical models, or information in the literature, is described in detail in Section 7.4. This section also shows the high degree of margin between anticipated and modeled failure rates and radionuclide releases.

8.3.3 Mechanical Disruption of Engineered Barriers (Section 2.2.1.3.2.3)

Acceptance Criteria 1–System Description and Model Integration are Adequate

The pertinent portions of this criterion include the description of the physical phenomena involved in the disruption process, assumptions and inputs used, and appropriate environmental conditions.

The physical phenomena are discussed in Sections 6.1 and 6.2.3. The assumptions are provided in Section 5 and inputs are provided in Section 4.1. The range of environmental conditions expected is discussed in Sections 6.1 and 6.2.3.

Acceptance Criteria 2–Data are Sufficient for Model Justification

The pertinent portion of the criterion include only the adequacy and use mechanical disruption data.

The source of the input data is provided in Section 4.1, while Section 6.2.3 describes how those data are used for the development of the model.

Acceptance Criteria 3–Data Uncertainty is Characterized and Propagated Through the Model Abstraction

The pertinent portions of the criterion include the model parameters, ranges, distributions, assumptions, and uncertainties.

The model parameters, ranges, distributions and uncertainties are discussed in Section 6.2.3. Assumptions are described in Section 5.

Acceptance Criteria 4–Model Uncertainty is Characterized and Propagated Through the Model Abstraction

The pertinent portions of the criterion include the use of alternative modeling approaches and the consideration of features, events and processes.

The alternative modeling approaches are discussed in Section 6 and the alternative conceptual models are discussed in Section 6.3. Features, events, and processes are discussed in Section 6.4.

Acceptance Criteria 5–Model Abstraction Output is Supported by Objective Comparisons

The pertinent portions of the criterion include the consistency of the results with empirical observations and the utilization of well documented procedures to conduct the tests.

The consistency of the results with empirical observations is discussed in Section 7.4.2 and the procedures utilized to conduct the tests are noted, where applicable, in the discussion in Sections 4 and 6.2.3.

8.3.4 Radionuclide Release Rates And Solubility Limits (Section 2.2.1.3.4.3)

Acceptance Criteria 1–System Description and Model Integration are Adequate

The pertinent portions of this criterion include only the need to reasonably account for the range of environmental conditions expected inside breached waste packages.

Environmental conditions expected inside breached waste packages are discussed in Section 6.1.

Acceptance Criteria 2–Data are Sufficient for Model Justification

> *The pertinent portions of the criterion include the adequacy of the input data and how the data are used.*

The data inputs are described in Section 4.1 and the use of the data is described in Section 6.2.

Acceptance Criteria 3–Data Uncertainty is Characterized and Propagated Through the Model Abstraction

> *The pertinent portions of the criterion include the model parameters, ranges, distributions, assumptions, and uncertainties.*

The model parameters, ranges, distributions and uncertainties are discussed in Section 6.2. Assumptions are described in Section 5.

Acceptance Criteria 4–Model Uncertainty is Characterized and Propagated Through the Model Abstraction

> *The pertinent portions of the criterion include the use of alternative modeling approaches and the consideration of features, events and processes.*

The alternative modeling approaches are discussed in Section 6 and the alternative conceptual models are discussed in Section 6.3. Features, events, and processes are discussed in Section 6.4.

Acceptance Criteria 5–Model Abstraction Output is Supported by Objective Comparisons

> *The pertinent portions of the criterion include the consistency of the results with empirical observations and the utilization of well documented procedures to conduct the tests.*

The consistency of the results with empirical observations is discussed in Section 7.4 and the procedures utilized to conduct the tests are noted, where applicable, in the discussion in Sections 4 and 6.1.

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